NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.7.4 RADIOLOGICAL CONSEQUENCES OF FUEL HANDLING ACCIDENTS

REVIEW RESPONSIBILITIES

Primary - Accident Evaluation Branch (AEB)

Secondary - Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

This SRP section covers the review of the radiological consequences of a postulated fuel handling accident. The purpose of the review is to evaluate the adequacy of system design features and plant procedures provided for the mitigation of the radiological consequences of accidents that involve damage to spent fuel. Such accidents include the dropping of a single fuel assembly and handling tool or of a heavy object onto other spent fuel assemblies. Such accidents may occur inside the containment, along the fuel transfer canal, and in the fuel building. The review includes the following:

1. The review is concerned with the selection of values of plant parameters for use in calculating the radiological consequences of a fuel handling accident, and the selection of the dose computation model, including assumptions of transport mechanisms and rates from the fuel handling area to the atmosphere, breathing rates, dose conversion factors, and other data that may affect the calculated dose.

2. The calculated doses are compared with the appropriate exposure guidelines to determine the acceptability of the exclusion area boundary and low population zone (LPZ) boundary and to confirm the adequacy of the engineered safety features (ESF) provided for the purpose of mitigating potential accident doses.

3. The containment ventilation system is reviewed with respect to its function as a dose mitigating engineered safety feature (ESF) system for a fuel handling accident inside the containment, including the radiation detection system on the containment purge/vent lines for those plants that will vent or purge the containment during fuel handling operations. The closure times

for the isolation valves in the lines are reviewed by the Containment Systems Branch (CSB).

4. The Effluent Treatment Systems Branch (ETSB) reviews, under SRP Section 6.5.1, the ESF atmosphere clean-up systems used to mitigate the radiological consequences of accidents. ETSB provides the filter efficiencies for the ESF systems to AEB for use in the analysis of the radiological consequences. This is a secondary review effort by ETSB.

5. Auxiliary Systems Branch (ASB) reviews under SRP Section 9.4.2 the design and operation of the spent fuel pool area ventilation system. The AEB reviewer verifies with the ASB the assumptions for the system with respect to its function as a dose mitigating system. This is a coordinating review function.

6. The movement of heavy loads (i.e., loads heavier than the combined weight of a spent fuel assembly and the fuel handling tool) or of irradiated fuel in the spent fuel pool and over the open reactor vessel is reviewed by ASB under SRP Sections 9.4.1 and 9.4.2. An analysis of the radiological consequences may be required for such drops of heavy objects if more than one fuel assembly can be damaged. The need for such calculation is determined by ASB who will advise AEB (note: the radiological consequences of a fuel cask drop in which the fuel inside the cask is damaged is reviewed by the AEB under SRP Section 15.7.5).

II. ACCEPTANCE CRITERIA

The AEB acceptance criteria for this SRP section are based on requirements of 10 CFR Part 100 (Ref. 1) with respect to the calculated radiological consequences of a fuel handling accident and General Design Criterion 61 (Ref. 2) with respect to appropriate containment, confinement, and filtering systems. Specific criteria necessary to meet the requirements are:

1. The plant site and dose mitigating ESF systems are acceptable with respect to the radiological consequences of a postulated fuel handling accident if the calculated whole-body and thyroid doses at the exclusion area and low population zone boundaries are well within the exposure guideline values of 10 CFR Part 100, paragraph 11. "Well within" means 25 percent or less of the 10 CFR Part 100 exposure guideline values, i.e., 75 rem for the thyroid and 6 rem for the whole-body doses.

2. The radioactivity control features of the fuel storage and handling systems inside containment and in the fuel building are acceptable if they meet the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," (Ref. 2) with respect to appropriate containment, confinement and filtering systems.

3. The model for calculating the whole-body and thyroid doses is acceptable if it incorporates the appropriate conservative assumptions in Regulatory Guide 1.25 (Ref. 3) with the exception of the guidelines for the atmospheric dispersion factors (X/Q values). The acceptability of the X/Q values is determined under SRP Section 2.3.4.

4. An ESF grade atmosphere clean-up system is required for the spent fuel storage area to reduce the potential radiological consequences.

5. The containment design is acceptable with respect to a postulated fuel handling accident if it possesses the capability for prompt radiation detection by use of redundant radiation monitors and automatic isolation if fuel handling operations inside containment occur when the containment is open to the environment (i.e., with a containment purge exhaust system). An acceptable alternative approach is containment venting through an ESF atmosphere cleanup system or containment isolation during fuel handling operations.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes specific aspects of this SRP section as are appropriate for the particular plant. The judgment on which areas need to be given attention and emphasis are determined by the similarity of the information presented by the applicant to that recently reviewed on other plants and whether items of special safety significance are involved.

1. The relevant portion of Chapter 15 of the applicant's safety analysis report (SAR) are reviewed to determine the values of those fuel parameters which affect fission product release and fuel pool iodine decontamination factors, including the maximum fuel rod pressurization, peak linear power density for the highest power assembly, maximum centerline operating fuel temperature for the peak assembly, average burnup for the peak assembly, and minimum water depth between the top of any damaged fuel rods and the water surface.

2. The staff performs an independent dose calculation using the assumptions in Regulatory Guide 1.25. If the values proposed by the applicant for gap activity or peak assembly power are less than those in Regulatory Guide 1.25 the Core Performance Branch (CPB) should be requested to review these values in a coordinated review effort. If other factors less conservative than those recommended in Regulatory Guide 1.25 are used, Reference 4 should be consulted to determine if an adequate basis for the proposed deviation exists.

   Three important parameters affecting the radiological consequences of a fuel handling accident are not covered in Regulatory Guide 1.25. These are the reactor design (stretch) power level, the earliest time after reactor shutdown that fuel handling operations can commence, and the number of fuel rods assumed to be damaged in a fuel handling accident. The reactor design power level is obtained from Section 1.1 or Chapter 15 of the SAR.

   Unless the applicant proposes otherwise, the standard technical specification (STS) values for minimum time to fuel handling are used to determine the earliest time after shutdown for fuel handling. (Current STS values are 24 hours for a boiling water reactor, 72 hours for a CE design and 100 hours for other pressurized water reactors). The applicant should provide in the SAR conservative analyses of the number of rods assumed damaged both for the spent fuel storage area and inside containment, and the Mechanical Engineering Branch (MEB) should be requested to verify the number of rods assumed damaged. Reference 6 may also be consulted in this regard.

3. Fuel handling accident in fuel buildings: The applicant's SAR is examined to assure that an ESF atmospheric cleanup system is included in the

design of the fuel storage facility to mitigate the radiological consequences of a fuel handling accident. Verification of acceptability and efficiencies of the atmosphere cleanup system are provided by the ETSB through the review of SRP Section 6.5.1. The reviewer should examine those pertinent aspects of the accident, especially with regard to the operational modes of the ventilation systems and location and response time of the radiation detectors to assure that any accidental release will be detected in sufficient time to be appropriately ducted and exhausted via ESF filters.

4. Fuel handling accident inside containment: The systems to mitigate the consequences are reviewed. If an applicant proposes that fuel handling will occur only when the containment is isolated, no radiological consequences need be calculated. If fuel handling operations occur only when the containment is exhausted to the environment via an ESF filter system, the radiological consequences should be calculated giving appropriate credit for this system. If the containment will be open during fuel handling operations, as with a containment purge exhaust system, the reviewer should verify that a prompt radiation detection and automatic containment isolation capability are provided and that the resulting doses are within the acceptance criteria given in subsection II.1 above.

   For a plant design with the containment open during the fuel movements, a review should be made of the applicant's analysis. This should include an examination of the type, location and redundancy of the radiation monitors intended to detect an activity release inside the containment and verification that detection is followed by automatic containment isolation. The reviewer should assess the time required to isolate the containment. This should include the instrument line sampling time (where appropriate), detector response time and containment purge isolation valve actuation and closure time. The containment is considered isolated only when the purge isolation valves are fully closed. The applicant's analysis should be reviewed regarding the travel time of any activity release starting from its release point above the refueling cavity or transfer canal and including travel time in ducts or ventilation systems up to the inner containment purge isolation valve.

   The time required for the release to reach the inner isolation valve is compared to the time required to isolate the containment. If the time required for the release to reach the isolation valve is longer than the time required to isolate containment, then essentially no release to the atmosphere occurs, and the reviewer's assessment should reflect this. If the time required for the release to reach the isolation valve is less than that required to isolate containment, and no mixing or dilution credit can be given, the reviewer should assume that the entire activity release escapes from the containment in evaluating the consequences. Claims for credit for dilution or mixing of a release due to natural or forced convection inside containment are reviewed and assessed. References 4 and 5 should be consulted and used by the reviewer for guidance in estimating dilution and mixing. Where mixing and dilution can be demonstrated within containment, the radiological consequences will be reduced by the degree of mixing and dilution occurring prior to containment isolation.

5. The atmospheric dispersion factors; X/Q values, to be used in analyzing the consequences of the accident are provided by the assigned meteorologist.

6. The doses calculated by the applicant and independently by the staff are compared to the acceptance criteria in subsection II. If the results of the dose calculations indicate the dose guideline values may be exceeded, alternatives which would reduce the doses to an acceptable level are examined and explored with the applicant (e.g., increased distance, better filters).

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided by the applicant and the staff independent dose calculations support conclusions of the following type, to be included in the staff's safety evaluation report:

> The staff finds that the applicant has provided an adequate system to mitigate the radiological consequences of a postulated fuel handling accident inside the containment and in the fuel building. The staff concludes that the fuel handling system meets the relevant requirements of General Design Criterion 61. The staff further concludes that the distance to the exclusion area and to the low population zone boundaries for the (INSERT PLANT NAME) site, in conjunction with the operation of dose mitigating ESF and implementation of plant procedures, are sufficient to provide reasonable assurance that the calculated offsite radiological consequences of a postulated fuel handling accident are well within the 10 CFR Part 100 exposure guidelines.
>
> The staff's conclusion is based on (1) the staff's determination that the design features and plant procedures at the (INSERT PLANT NAME) facility meet the requirements of General Design Criterion 61 with respect to radioactivity control; (2) the staff review of the applicant's assumptions and analyses of the radiological consequences from the fuel handling accident; and (3) the staff's independent analyses using the assumptions in Regulatory Guide 1.25, Portions C.1.a through C.1.k.

V. IMPLEMENTATION

The following provides guidance to applicants and licensees regarding the staff's plans for using this SRP Section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guide.

VI. REFERENCES

1. 10 CFR Part 100, Paragraph 11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

2. 10 CFR Part 50, Appendix A, General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control."

3. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

4. Evaluation of Fission Product Release and Transport for a Fuel Handling Accident by G. Burley, Radiological Safety Branch, Division of Reactor Licensing, revised October 5, 1971.

5. Industrial Ventilation/A Manual of Recommended Practice - American Conference of Governmental Industrial Hygienists.

6. Long Island Lighting Co., et al., Docket No. STN 50-516/517, Further additional supplemental testimony on contention I.D.2 (Spent Fuel Handling Accident) by Walter L. Brooks, et al.